UPM Activities on Sensitivity and Uncertainty Analysis of Assembly Depletion

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Outline

1. Benchmark Assembly Depletion: II-2a

1.1 Objectives and requested output

1.2 Benchmark Specifications. Cases:

1.2.1 PWR Assembly Depletion

1.2.2 BWR Assembly Depletion

2. Conclusion and “On Going” Activities

1.1. Exercise II-2a: Objectives

- Objective of Exercise II-2a: To assess uncertainties along fuel assembly depletion (long-term time phenomena associated with fuel burnup)
- Uncertainties in:
  - ✓ K-inf, pin-power and burnup distributions
  - ✓ Homogenized two-group nodal cross sections
  - ✓ Number densities of selected nuclides
- Due to:
  - ✓ Uncertainties in cross-sections, decay constants, fission yields
  - ✓ Manufacturing uncertainties
  - ✓ Methodological uncertainties of time-dependent nuclide isotopics modeling

Requested Output

- The neutron multiplication factors, k_{inf} (assembly k_{inf}), and associated uncertainties at the 6 following burnup steps during the depletion: BOC, 0.2 GWd/t, 0.25*EOC, 0.50*EOC, 0.75*EOC, EOC
- The homogenized two-group nodal cross-sections (with 0.625 eV as the thermal cut-off energy) as well as the associated uncertainties and covariance matrix at the same 6 burnup steps. The format is provided in Figure 48.

```
* -----------------------------------------------------------
* BURNUP 0.00
* -----------------------------------------------------------
*
* Transport XSEC Table
  GROUP 1, GROUP 2
* Diffusion coefficient Table
  GROUP 1, GROUP 2
* Absorption XSEC Table
  GROUP 1, GROUP 2
* Nu-Fission XSEC Table
  GROUP 1, GROUP 2
* Kappa-Fission XSEC Table
  GROUP 1, GROUP 2
* Scattering XSEC Table
  GROUP 1 -> 1, GROUP 1 -> 2
  GROUP 2 -> 1, GROUP 2 -> 2
* ADF Table
  GROUP 1, GROUP 2
* Fission Spectrum
  CHI(G1) CHI(G2)
* Inverse Velocity
  IVEL(G1) IVEL(G2)
* Delay Neutron Decay Constant (Lambda)
  LAMBDA(G1) LAMBDA(G2) LAMBDA(G3) LAMBDA(G4) LAMBDA(G5) LAMBDA(G6)
* Delay Neutron Fraction (Beta)
  BETA(G1) BETA(G2) BETA(G3) BETA(G4) BETA(G5) BETA(G6)
*
* -----------------------------------------------------------
* BURNUP 0.20
* -----------------------------------------------------------
....
....
```

Figure 48: Exercise II-2 Depletion Case Cross-Sections Template

Requested Output

- Pin power distribution and burnup of each fuel pin in the unit of GWd/t and associated uncertainties at the same 6 burnup values.

- Number densities and associated uncertainties of all nuclides specified in Table 119 (14 actinides and 31 FPs), which are averaged over all fuel rods, in the unit of $10^{24}/cm^3$ at EOC (not at any other depletion time steps).

Table 119: Exercise II-2 Depletion Case Nuclides List

Actinides	U-234, 235, 236, 238 Np-237 Pu-238, 239, 240, 241, 242 Am- 241, 243 Cm-242, 244
FPs	Sr-90 Mo-95 Tc-99 Ru-101 Rh-103 Ag-109 I-129 Xe-131, 135 Cs-133, 134, 137 Ce-144 Nd-142, 143, 144, 145, 146, 148 Sm-147, 149, 150, 151, 152 Eu-153, 154, 155 Gd-155, 156, 157, 158

1.2.1 CASE: PWR Assembly. Specification

- PWR TMI-1 assembly design: 15x15 lattice array with two types of fuel pins: UO2 pins to 4.12 w/o and UO2-Gd2O3 rods

	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
1	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2	-	g	-	-	-	-	-	-	-	-	-	-	-	g	-
3	-	-	-	-	-	G	-	-	-	G	-	-	-	-	-
4	-	-	-	G	-	-	-	-	-	-	-	G	-	-	-
5	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
6	-	-	G	-	-	G	-	-	-	G	-	-	G	-	-
7	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
8	-	-	-	-	-	-	-	I	-	-	-	-	-	-	-
9	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
10	-	-	G	-	-	G	-	-	-	G	-	-	G	-	-
11	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
12	-	-	-	G	-	-	-	-	-	-	-	G	-	-	-
13	-	-	-	-	-	G	-	-	-	G	-	-	-	-	-
14	-	g	-	-	-	-	-	-	-	-	-	-	-	g	-
15	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-

Figure 19: TMI-1 FA Pin Layout

1.2.1 CASE: PWR Assembly. Specification

- Core boundary conditions

Table 50: TMI-1 Core Boundary Conditions

Core Power	2772 MWt
Coolant Temperature	578 K
Core Pressure	15.51 MPa
Core Coolant Flow Rate	16052.4 kg/sec

- Irradiation at HFP conditions (given in Figure 4 of Vol. I)

Parameter / Reactor condition	HZP	HFP
Fuel temperature, [K]	551	900
Cladding temperature, [K]	551	600
Moderator (coolant) temperature, [K]	551	562
Moderator (coolant) density, [kg/m^3]	766	748.4
Reactor power ,[MWt]	2.772	2 772

- Irradiation history: constant power over the irradiation period

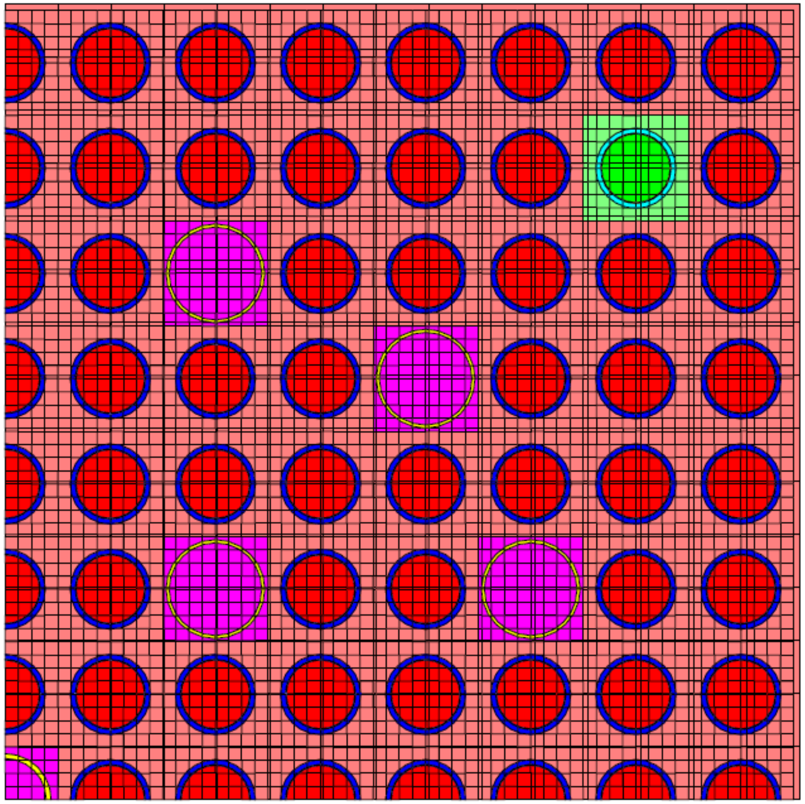
Table 52: TMI-1 Irradiation History

Time	Power
Days	MW/t
875	45.00

PWR: SCALE6.1.1 Calculation methodology for depletion

- TRITON multipurpose sequence of SCALE6.1.1 to perform coupled xs processing, transport and depletion
- **Cross-section libraries**: SCALE 238g xs library from ENDF/B-VII.0 (v7-238); TRITON's `parm=weight` option used to collapse the 238g master library to a 49g problem-dependent library
- **Cross-section processing**: CENTRM LATTICECELL calculations; one latticecell per fuel type ⇒ 2 different latticecells blocks
- **Transport calculations**: S_N=10, convergence criteria set at 1.0E-5, fine mesh of 4x4 for the square-pitched units, CMFD acceleration (`cmfd=rect`, `cmfd2g=yes`)
- **Depletion calculations:**
  - ✓ `addnux=4` (adding all nuclides in the ENDFB-B/VII library)
  - ✓ 2 fuel mixtures depleted: UO2 fuel rods and Gd rods
  - ✓ Normalization using total system power for non-Gd rods and flux for Gd rods
  - ✓ 15 depletion intervals: 0.4 GWd/t, steps of ~2.0 GWd/t until an assembly-average burnup of 20 GWd/t, steps of ~ 2.25 GWd for larger burnups

PWR: SCALE6.1.1 Calculation methodology for depletion



- ¼ FA
- Two fuel mixtures to be depleted:
  - ✓1 fuel material for UOx pins
  - ✓1 fuel materials for Gd pins

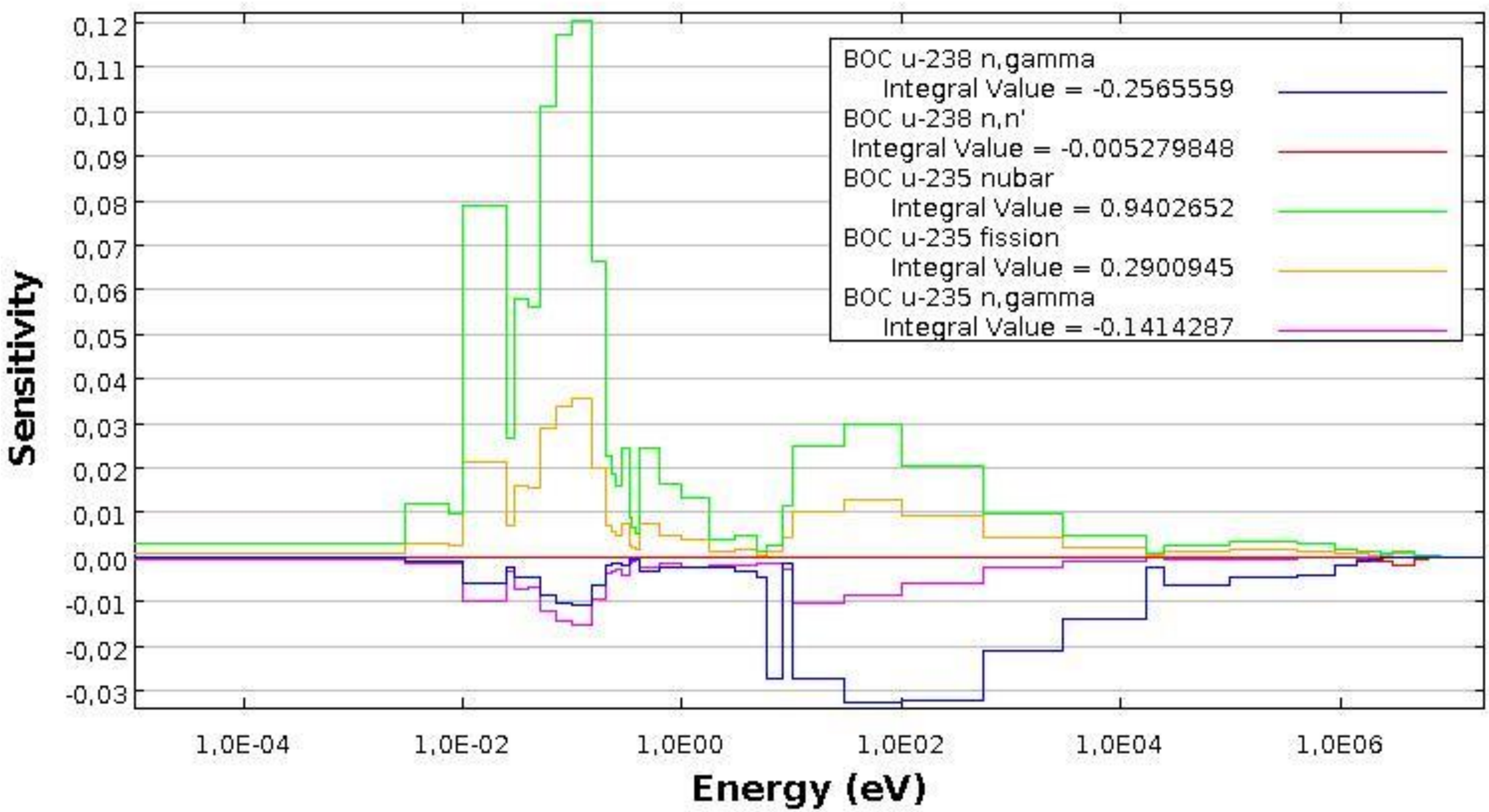
Calculation methodology for uncertainty

- **Uncertainty propagation in criticality along burnup:**
  - ✓ TSUNAMI-2D / SCALE6.1.1
  - ✓ SAMS module to compute sensitivities of keff values to nuclear data (and propagate uncertainties)
  - ✓ TSAR module combines previous sensitivities and produces sensitivities used to propagate nuclear data uncertainties
  - ✓ Nuclear data uncertainties taken from SCALE6.1.1/COVA

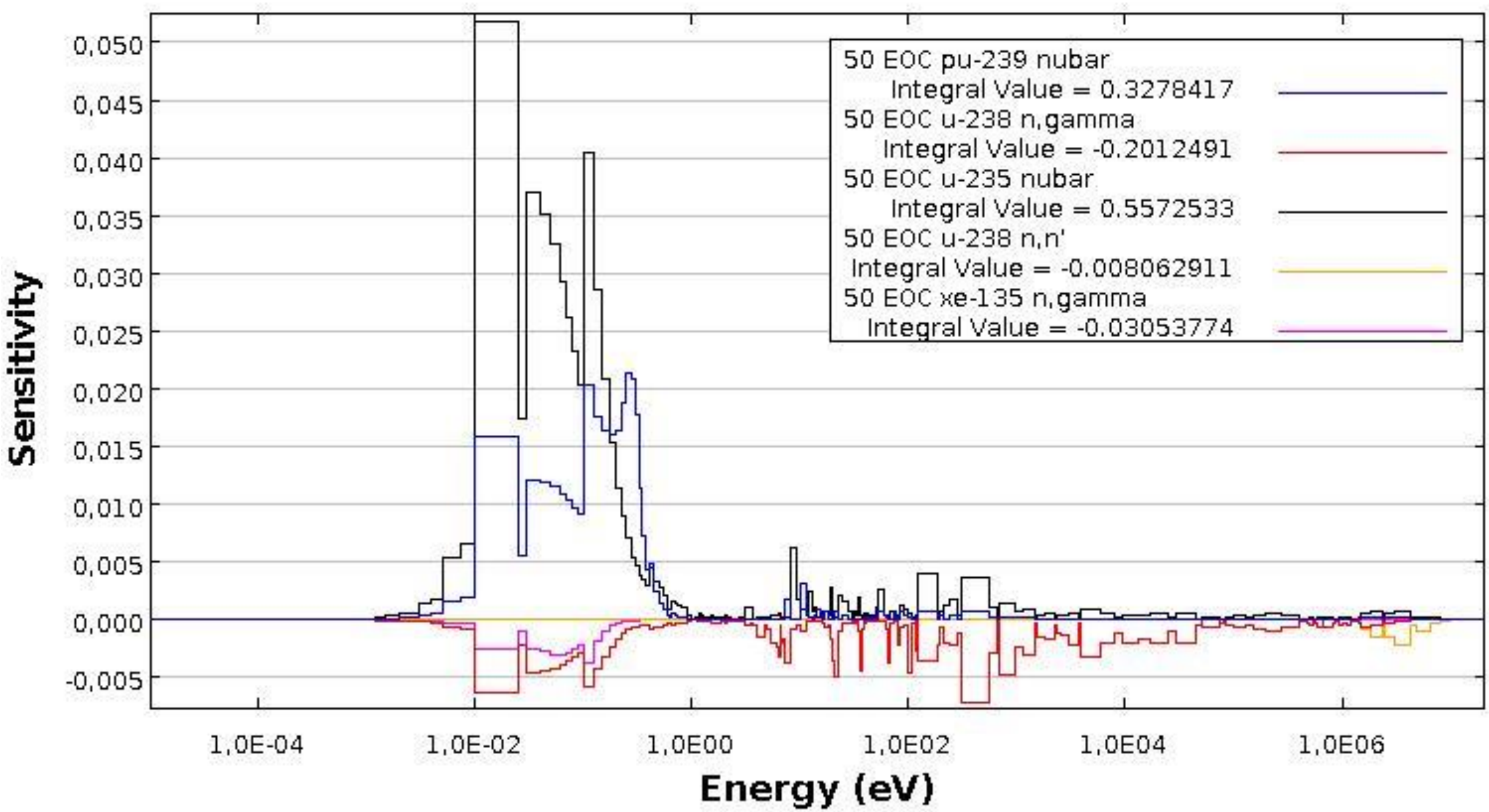
PWR: SCALE6.1.1 Results for depletion calculation

BOC			**0.50*EOC=MOC**			**EOC**	
	uncertainty		K-eff	uncertainty		K-eff	uncertainty
K-eff	% dk/k			% dk/k			% dk/k
1.36087	0.514		1.14509	0.578		0.98519	0.712
Contributors	**% dk/k**		**Contributors**	**% dk/k**		**Contributors**	**% dk/k**
U-238 n,gamma	0.327		Pu-239 nubar	0.338		Pu-239 nubar	0.478
U-235 nubar	0.269		U-238 n,gamma	0.260		U-238 n,gamma	0.268
U-235 n,gamma	0.192		U-235 nubar	0.159		U-238 n,n'	0.195
U-235 n,gamma	0.112		U-238 n,n'	0.154		Pu-239 fission	0.190
U-238 n,n'	0.102		Xe-135 n,gamma	0.127		Pu-239 n,gamma	0.160
U-235 fission	0.091		Pu-239 fission	0.119		Xe-135 n,gamma	0.118

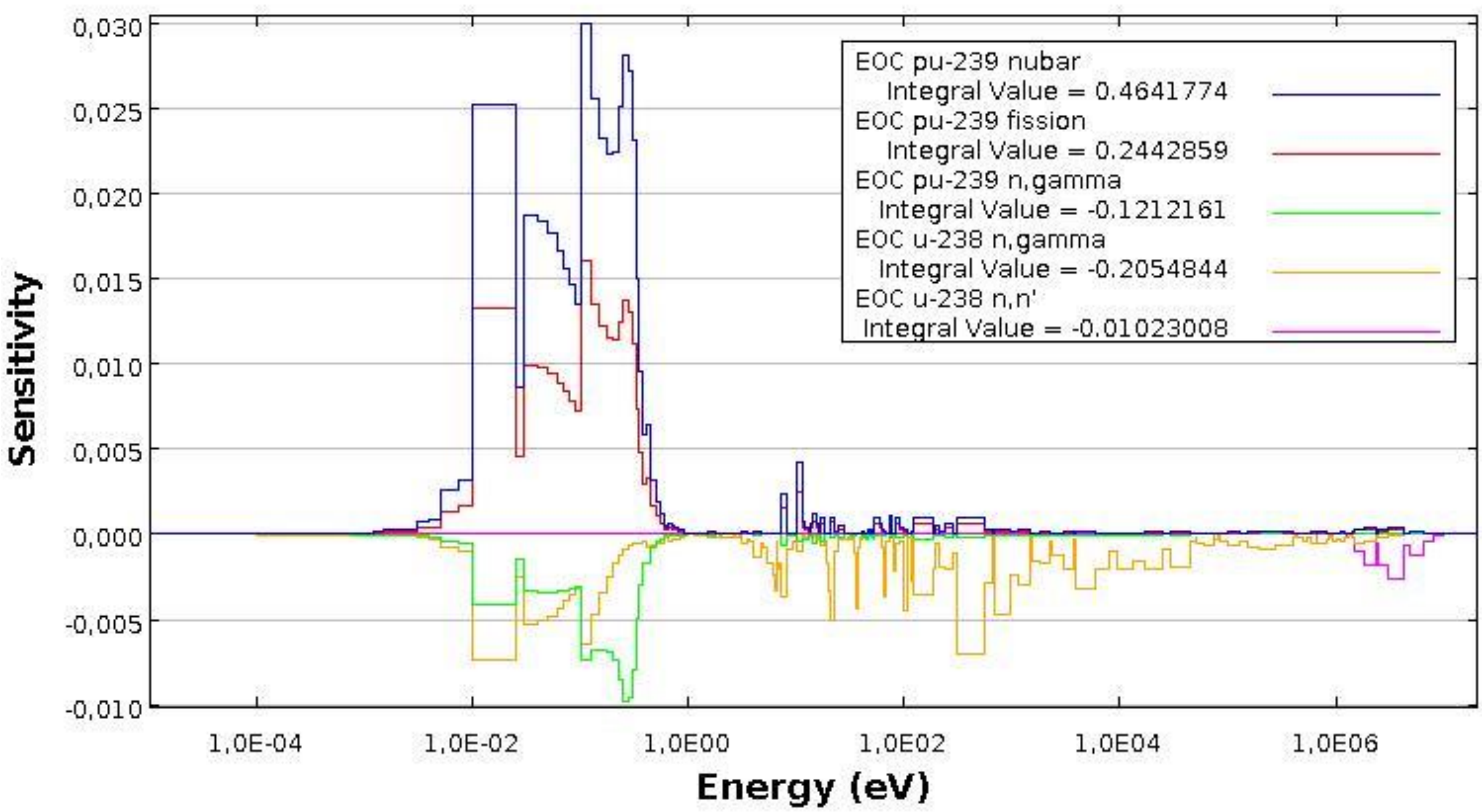
PWR: SCALE6.1.1, Keff Sensitivity Analysis at BOC



PWR: SCALE6.1.1, keff Sensitivity Analysis at MOC



PWR: SCALE6.1.1, keff Sensitivity Analysis at EOC



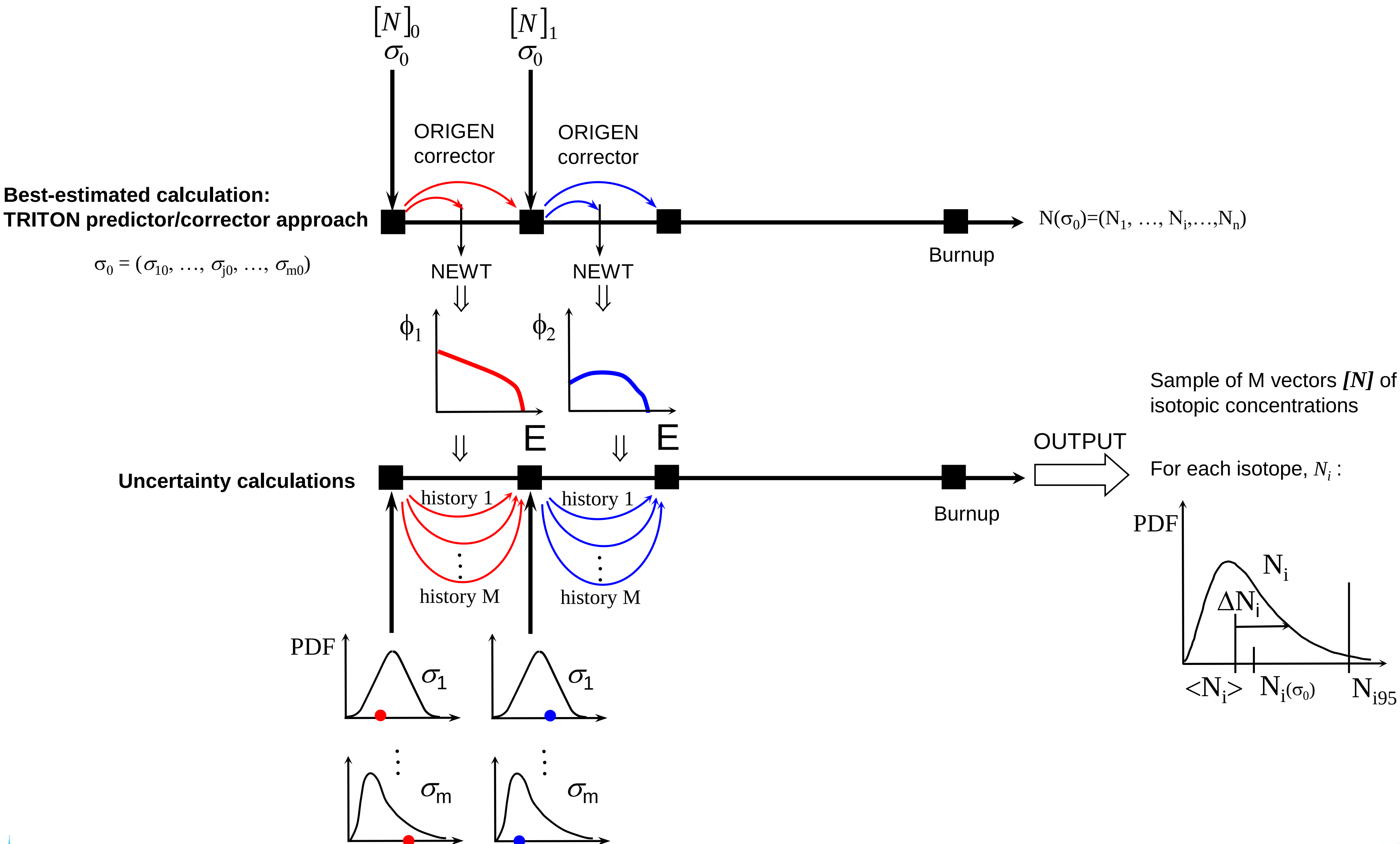
PWR: SCALE6.1.1 Results for depletion calculation

	BOC			0.50*EOC=MOC			EOC	
	Response	Uncertainty (%)		Response	Uncertainty (%)		Response	Uncertainty (%)
Transport Fast	0.22	-		0.22	-		0.22	-
Transport Thermal	0.97	-		0.94	-		0.95	-
Difusion F	1.49	-		1.49	-		1.49	-
Difusion T	0.34	-		0.35	-		0.35	-
Absorption F	0.01	0.87		0.01	0.87		0.01	0.88
AbsorptionT	0.10	0.22		0.11	0.24		0.10	0.27
Nu-fission F	0.01	0.59		0.01	0.79		0.01	1.14
Nu-fission T	0.16	0.45		0.16	0.56		0.13	0.71
Scat. FF	0.52	0.82		0.53	0.88		0.53	0.88
Scat. TT	1.31	0.13		1.31	0.15		1.31	0.15
Scat. FT	0.02	0.81		0.02	1.21		0.02	1.20

Calculation methodology for uncertainty: Hybrid Method

- **Uncertainty propagation in number densities along burnup:**
  - ✓ The Monte Carlo Hybrid Method was used to account for the impact of uncertainties in the basic nuclear data (cross-section, decay data and fission yields) along the consecutive spectrum-depletion steps
  - ✓ Best-estimated calculation: TRITON predictor/corrector approach
  - ✓ Uncertainty calculation: simultaneous random sampling of the PDF of all the input parameters

Calculation methodology for uncertainty: Hybrid Method



Uncertainty

Isotope	0.2 GWd/TMU	.25*EOC	MOC	.75*EOC	EOC
U234	0.0	1.0	1.9	2.9	3.6
U235	0.0	0.1	0.3	0.4	0.5
U236	1.2	1.2	1.2	1.2	1.2
U238	0.0	0.1	0.1	0.2	0.2
Np237	13.0	4.6	3.6	3.3	3.2
Pu238	11.4	5.6	3.9	3.1	2.7
Pu239	1.4	1.4	1.4	1.4	1.4
Pu240	2.7	1.8	1.7	1.7	1.7
Pu241	3.2	1.8	1.7	1.7	1.7
Pu242	3.8	2.1	2.2	2.6	3.0
Am241	3.7	1.9	2.0	2.2	2.4
Am243	**10.1**	**9.0**	**8.6**	**8.1**	**7.6**
Cm242	4.9	3.0	2.7	2.3	2.1
Cm244	**10.5**	**9.3**	**8.9**	**8.5**	**8.1**

Isotope	0.2 GWd/TMU	.25*EOC	MOC	0.75*EOC	EOC
Sr90	0.3	0.4	0.5	0.6	0.7
Mo95	0.3	0.5	0.7	0.8	0.9
Tc99	0.3	0.7	0.9	1.0	1.1
Ru101	0.3	0.8	0.9	1.1	1.1
Rh103	0.3	1.0	1.3	1.5	1.7
Ag109	0.7	1.5	1.6	1.7	1.8
I129	0.3	0.9	1.1	1.2	1.2
Xe131	0.3	0.9	1.2	1.6	2.0
Xe135	3.4	3.5	3.6	3.7	3.8
Cs133	0.3	0.7	1.0	1.1	1.3
Cs134	4.7	4.7	4.6	4.5	4.4
Cs137	0.3	0.7	0.9	1.0	1.0
Ce144	0.3	0.6	0.8	0.9	1.0
Nd142	4.4	4.7	4.7	4.7	4.7
Nd143	0.3	0.6	1.0	1.2	1.5
Nd144	0.7	1.0	1.2	1.3	1.4
Nd145	0.3	0.7	1.0	1.3	1.6
Nd146	0.3	0.7	1.0	1.3	1.5
Nd148	0.3	0.8	1.0	1.1	1.2
Sm147	0.3	0.8	1.3	1.7	2.0
Sm149	0.8	1.9	2.2	2.5	2.8
Sm150	1.1	0.9	1.3	1.7	2.0
Sm151	0.4	2.1	2.5	2.8	3.0
Sm152	0.5	1.2	1.6	2.1	2.4
Eu153	0.3	1.5	2.0	2.3	2.6
Eu154	**4.5**	**5.3**	**6.2**	**6.9**	**7.4**
Eu155	**0.5**	**3.9**	**4.8**	**5.1**	**4.9**
Gd155	1.1	4.4	3.9	4.1	4.1
Gd156	0.3	0.4	0.9	1.6	2.2
Gd157	1.4	4.5	4.0	4.5	5.3
Gd158	0.0	0.3	0.6	1.0	1.4

1.2.2 CASE: BWR Assembly. Specification

- BWR PB-2 assembly design: 7x7 lattice array with six pin types : 4 types of UO2 pins with different fuel enrichments from 1.33 w/o to 2.93 and 2 types of UO2-Gd2O3 rods
- Cladding thickness changed in Volume II (Specification for the Core Cases – Phase II) with respect to Volume I (Spec. for the Neutronics Cases – Phase I) → assumed specifications of Vol. II

WIDE-WIDE CORNER

4	3	3	2	2	2	3
3	2	1	1	1	1	2
3	1	5A	1	1	5A	1
2	1	1	1	1	1	1
2	1	1	1	6B	1	1
2	1	5A	1	1	1	2
3	2	1	1	1	2	2

Table 42: PB-2 Fuel Rod Dimensions and Parameters

Cladding OD	14.30 mm	
Cladding ID	13.36 mm	
Cladding Thickness	0.47 mm	(0.9398 mm in Vol. I)
Pin Pitch	18.75 mm	
Fuel Pellet OD	12.12 mm	
Fuel Pellet Height	10.7 mm	
% Density	95.1% TD	

BWR Assembly. Specification

- Core boundary conditions: 574.8 K?? (saturation temperature at core pressure of 7.14 MPa ~ 560K)

Table 43: PB-2 Core Boundary Conditions

Core Power	3293 MWt
Coolant Temperature	574.8 K
Core Pressure	7.14 MPa
Coolant Flow Rate	12915 kg/s
Inlet Enthalpy	1213 kJ/kg
Average Void Fraction	40%

- Assuming irradiation at HFP conditions (given in Figure 3 of Vol. I) No specification given outside-channel ⇒ assumed with 0% V.F. out-channel (dens=0.74308 g/cm^3)

Parameter / Reactor condition	HZP	HFP
Fuel temperature, [K]	552.833	900
Cladding temperature, [K]	552.833	600
Moderator (coolant) temperature, [K]	552.833	557
Moderator (coolant) density, [kg/m^3]	753.978	460.72
Reactor power ,[MWt]	3.293	3 293
Void fraction (%)	-	40

- Irradiation history: constant power over the irradiation period

Table 45: PB-2 Irradiation History

Time	Power
Days	MW/t
1400	32.00

BWR: SCALE6.1.1 Calculation methodology for depletion

- TRITON multipurpose sequence of SCALE6.1 to perform coupled xs processing, transport and depletion

- **Cross-section libraries**: SCALE 238g xs library from ENDF/B-VII.0 (v7-238); TRITON's parm=`weight` option used to collapse the 238g master library to a 49g problem-dependent library

- **Cross-section processing**:
  - ✓ For Gd-bearing fuel rods:
    - ✓ MULTIREGION treatment with 5 equal-volume rings
  - ✓ For UO2 fuel rods: CENTRM latticecell treatment
    - ✓ Dancoff factors computed using MCDANCOFF module
    - ✓ TRITON's `assign` option used to lump rods for resonance processing purposes along depletion. Fuel lumping strategy:
      - Pins with same enrichment
      - Pins with similar Dancoff factor

BWR: SCALE6.1.1 Calculation methodology for depletion

- Dancoff factor map: accurate Dancoff factors computed with MCDancoff module

WIDE-WIDE CORNER

4	3	3	2	2	2	3
3	2	1	1	1	1	2
3	1	5A	1	1	5A	1
2	1	1	1	1	1	1
2	1	1	1	6B	1	1
2	1	5A	1	1	1	2
3	2	1	1	1	2	2

0.270						
0.363	0.475					
0.364	0.479	Gd			Gd	
0.364	0.479	0.482	0.482			
0.364	0.480	0.481	0.482	Gd		
0.363	0.477	Gd	0.480	0.479	0.478	
0.280	0.376	0.377	0.378	0.377	0.376	0.290

Same enrichment, different Dancoff factors ⇒ 2 CENTRM latticecells

Same enrichment, similar Dancoff factors ⇒ 1 CENTRM latticecell

- A total of 9 different CENTRM latticecells for the fuel assembly

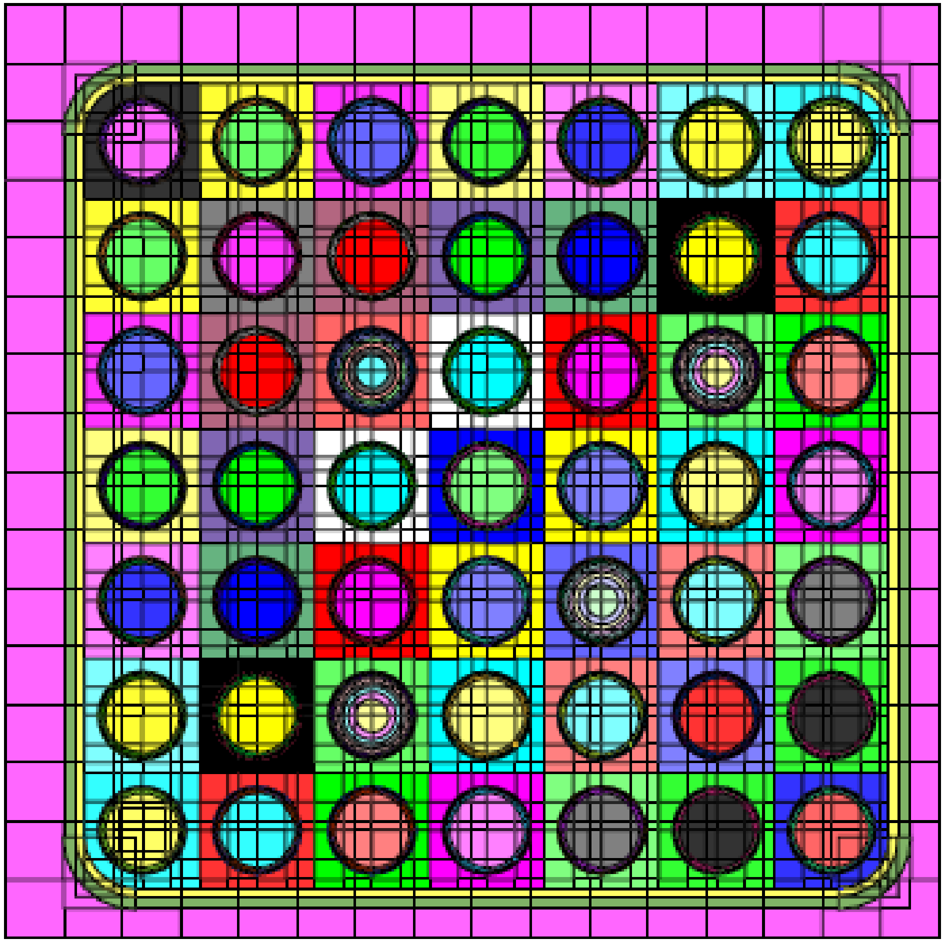
BWR: SCALE6.1.1 Calculation methodology for depletion

- **Transport calculations**: S_N=10, convergence criteria set at 1.0E-5, fine mesh of 4x4 for the square-pitched units, CMFD acceleration (`cmfd=rect`, `cmfd2g=yes`)

- **Depletion calculations:**

  - ✓ addnux=4 (adding all nuclides in the ENDFB-B/VII library)
  - ✓ All fuel mixtures in the fuel assembly were depleted individually
  - ✓ Normalization using total system power for non-Gd rods and flux for Gd rods
  - ✓ 25 depletion intervals: 0.4 GWd/t, steps of ~1.5 GWd/t until an assembly-average burnup of 20 GWd/t, steps of ~ 2.25 GWd for larger burnups

BWR Assembly modeling



- FA diagonally-symmetric
- A separate mixture number specified for each region to be depleted:
  - ✓25 fuel materials for UOx pins
  - ✓15 fuel materials for Gd pins (3 different pins * 5 regions)
  - ✓40 materials irradiated

Summary and conclusions

- **Preliminary results for PWR depletion numerical benchmark with SCALE6.1.1.**

- BWR depletion numerical benchmark is still in progress

- Future work: Experimental Benchmark

  - PWR, Takahama-3 (TK-3)
  - BWR, Fukushima Daina-2 (FK-2)

Acknowledgements

This work is supported by:

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Case 4a: BWR, Depletion, Experimental

This test case is an experimental case designed off of the Fukushima Daina-2 (FK-2) irradiation and subsequent post-irradiation examination (PIE) performed at the Tokyo Electric Company's plant by the Japan Atomic Energy Research Institute (JAERI). The fuel assembly to be modeled was irradiated in this reactor where its power levels were monitored and recorded. After the irradiation, several fuel rods were examined to determine the concentrations of several important nuclides. The fuel assembly is an 8x8 BWR assembly with several gadolinium pins as well as two water rods. The geometry of the FK-2 FA is defined by Figure 27 [37].

	1	2	3	4	5	6	7	8
1	5	4	3	3	3	3	4	5
2	4	1	G	2	2	G	1	4
3	3	G	2	4	4	2	G	3
4	3	2	4	3	W	4	2	3
5	3	2	4	W	3	4	2	3
6	3	G	2	4	4	2	G	3
7	4	1	G	2	2	G	1	4
8	5	4	3	3	3	3	4	5

Figure 27: FK-2 FA Pin Layout

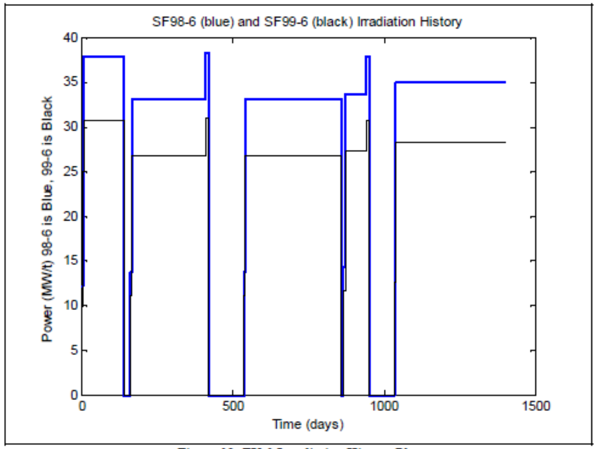


Figure 28: FK-2 Irradiation History Plot

The numbers in Figure 27 represent the various rods that are in the FA, and they are defined in Table 59. The orange and red rods indicate the location of the rods from which the PIE data were obtained.

Table 59: FK-2 FA Pin Descriptions

G	Gd_2O_3+UO_2 pin
W	water rod
1	3.63% ^{235}U pin
2	3.22% ^{235}U pin
3	3.18% ^{235}U pin
4	2.72% ^{235}U pin
5	1.89% ^{235}U pin
1	SF-98 Location
G	SF-99 Location

SF-98 and SF-99 are the two fuel pins that were analyzed in the PIE. The nuclides found in these two pins are available as experimental data with which to compare the calculated results. Other relevant geometrical details about the FA are provided in Table 60.

Depletion Experimental

Case 5a: PWR, Depletion, Experimental

Similar to the previous test case (from FK-2), this case was also a PIE performed by JAERI on an irradiated fuel assembly from the Takahama-3 (TK-3) PWR reactor. The geometry of the TK-3 FA is defined by Figure 29 [52].

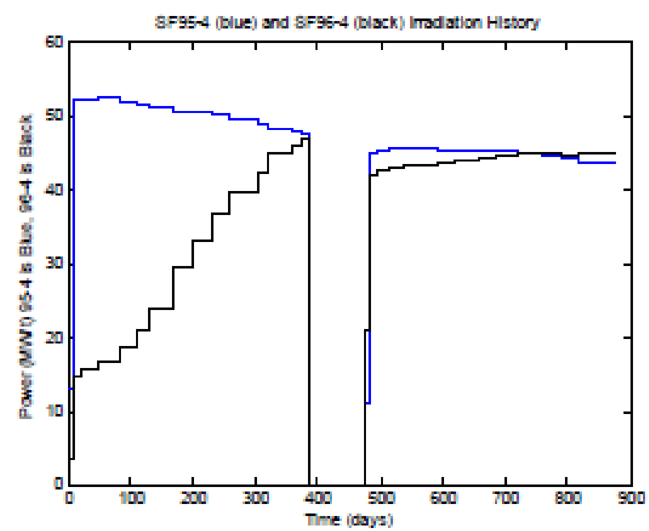
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
1	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
2	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
3	-	-	-	-	G	W	-	-	W	-	-	W	G	-	-	-	-
4	-	-	-	W	-	-	-	-	G	-	-	-	-	W	-	-	-
5	-	-	G	-	-	-	-	-	-	-	-	-	-	-	G	-	-
6	-	-	W	-	-	W	-	-	W	-	-	W	-	-	W	-	-
7	-	-	-	-	-	-	G	-	-	-	G	-	-	-	-	-	-
8	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
9	-	-	W	G	-	W	-	-	W	-	-	W	-	G	W	-	-
10	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
11	-	-	-	-	-	-	G	-	-	-	G	-	-	-	-	-	-
12	-	-	W	-	-	W	-	-	W	-	-	W	-	-	W	-	-
13	-	-	G	-	-	-	-	-	-	-	-	-	-	-	G	-	-
14	-	-	-	W	-	-	-	-	G	-	-	-	-	W	-	-	-
15	-	-	-	-	G	W	-	-	W	-	-	W	G	-	-	-	-
16	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-
17	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-	-

Figure 29: TK-3 FA Pin Layout

The numbers in the above figure represent the various rods that are in the FA, and they are defined in Table 65. The orange and red locations show the rods from which the PIE data were obtained.

Table 65: TK-3 FA Pin Descriptions

Marker	Rod Type
G	6.0 w/o Gd 2.63% ^{235}U pin
W	Control Rod (water filled)
-	4.11% ^{235}U fuel pin
-	SF-95 Location (NT3G23 FA)
G	SF-96 Location (NT3G23 FA)



Requested Output

Table 120: Exercise II-2 Depletion Experimental Case Results Template

Case #:	4a	
Case Type:	Depletion, Experimental	
Reactor Type:	BWR	
Experiment Name:	FK-2 PIE	
Scenario Time:	1402	days
Target Parameter:	Nuclide Concentrations	
Axial Location:	2050	mm
Nuclide:	Value	st dev.
U-234		
…		
Gd-158		
Burnup (MWd/MTU):		
K_{eff}:		